

LA-UR-12-25405

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Title:	ASME Code Case Development Strategy for Ferritic/Martensitic HT-9 Steel
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**Global Nuclear
Network Analysis, LLC**

ASME CODE CASE DEVELOPMENT STRATEGY FOR FERRITIC/MARTENSITIC HT-9 STEEL

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Report No. GNNA-12-025

19 September 2012

Los Alamos National Laboratory

EXECUTIVE SUMMARY

The US Department of Energy (DOE), Office of Nuclear Energy (NE), is supporting the development of an ASME Code Case for adoption of 12Cr-1Mo-VW ferritic/martensitic (F/M) steel, commonly known as HT-9, primarily for use in design of liquid-metal fast reactors (LMFR) and components. In 2011, Los Alamos National Laboratory (LANL) nuclear engineering staff began assisting in the development of a small modular reactor (SMR) design concept, previously known as the Hyperion Module, now called the Gen4 Module, being developed by Gen4 Energy, Inc. located in Denver, Colorado. LANL engineering staff had initially proposed either T91 (9Cr-1Mo-VNb) or HT-9 for the reactor vessel and components, as well as fuel clad and ducting. Although T91 is currently an approved/authorized material for elevated temperature design in the ASME Code, the preference is utilization of HT-9 due to its extensive irradiation database and similar creep resistance as T91. In addition, HT-9 has relatively good corrosion properties for fast reactor environment, such as liquid lead bismuth eutectic coolant.

Although the ASME material Code Case, for adoption of HT-9 as an approved elevated temperature material for LMFR service, is the ultimate goal of this project, several key deliverables have been accomplished in FY2012 to provide assurance to DOE/NE of the viability and feasibility of pursuing this goal. An important key deliverable was the recently completed, *Gap Analysis of Material Properties Data for Ferritic/Martensitic HT-9 (1)*. This report provided a cursory research, accumulation, and documentation of specific material parameters; physical, mechanical, and environmental, which are necessary for ensuring a successful ASME Code Case. Time-independent tensile and ductility data and time-dependent creep and creep-rupture behavior are some of the material properties required for a successful ASME Code case. Unfortunately, results of the Gap Analysis report show that many of the material properties have limited data availability in open-source documentation, or are contained in Applied Technology documents, or there is insufficient data to support the ASME requirement from 3-separate heats of material.

This report addresses the strategic activities necessary during FY2013 and beyond to ensure the successful development of an ASME Code Case for adoption of HT-9 for elevated temperature service. Specifically, the report addresses the following targeted areas required to be conducted and completed before a formal Code Case submittal package is delivered to ASME, Sec. III, Div. 1, *Nuclear Vessels*, and Sec. II, *Materials*:

- HT-9 Material database collection
- Completion of ASTM A771 updates
- Commercial reactor vendor/designer collaboration
- Commercial material manufacturer collaboration

- ASTM Subcommittee support for product-form specifications
- National laboratory subject matter expertise collaboration
- ASME Sec. II. Materials and Sec. III, Nuclear Vessels
- Materials research and development plan for HT-9
- DOE/NE funding availability

Table of Contents

1.0	INTRODUCTION	6
2.0	BACKGROUND	6
3.0	ASME CODE DATA REQUIREMENTS.....	6
4.0	STRATEGIC ENDEAVORS	8
4.1	Precursors	8
4.1.1	HT-9 Material Database Collection	8
4.1.2	Finalize ASTM A771 Update	9
4.2	Commercial Reactor Designer Collaboration	10
4.3	Commercial Material Manufacturer Collaboration.....	11
4.4	ASTM International Support.....	11
4.5	National Laboratories Support	12
4.6	ASME Collaborations	13
4.7	Material Research and Development Plan	14
5.0	DOE/NE FUNDING PLAN	17
6.0	CONCLUSIONS AND RECOMMENDATIONS	18
7.0	REFERENCES	19

List of Tables

Table 1 – ASTM A771 Alloy Composition Limits for Austenitic Stainless Steel Tubing	9
Table 2 – Typical Mechanical Properties Derived from Several Sources	10
Table 3 – ASTM Subcommittees Responsible for Product-Form	12

1.0 INTRODUCTION

Los Alamos National Laboratory (LANL) with assistance Global Nuclear Network Analysis, LLC (GNNA, LLC) has developed a gap analysis between American Society of Mechanical Engineers (ASME) requirements for elevated temperature design material properties and the US Department of Energy (DOE) database titled, *Nuclear Systems Materials Handbook (NSMH)* (2), for 12Cr-1Mo-VW ferritic/martensitic (F/M) steel HT-9. The gap analysis is one of the major milestones supporting the DOE Work Package at LANL.

2.0 BACKGROUND

The Small Modular Reactor (SMR) community has an immediate need for use of ferritic/martensitic (F/M) steel, commonly known as HT-9. The F/M steel, HT-9, is being considered not only for fuel-cladding but for reactor vessel components and piping.

The US Department of Energy (DOE), Office of Nuclear Energy (NE), is supporting the SMR designs in fostering advanced concepts and materials. Certain SMR designs will be utilizing liquid-metal cooled fast reactor (LMFR) systems that require materials with excellent high-temperature properties, corrosion resistance to the liquid metal and creep resistance. One such design is from Hyperion Power Generation Inc., which has recently changed its name to Gen4 Energy, Inc. Gen4 Module is a liquid lead-bismuth eutectic (LBE) cooled SMR with fuel pellets contained in HT-9 cladding tubes. Along with scientific and engineering support from Los Alamos National Laboratory (LANL) and funding from the DOE/NE, the Gen4 Energy Inc., engineering design staff are pursuing the use of HT-9 for its liquid-metal cooled fast reactor (LMFR).

However, in order to meet Nuclear Regulatory Commission (NRC) compliance for use of specific reactor pressure boundary materials, approval must be obtained from the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code, Section III, Division 1, “*Rules for Construction of Nuclear Facility Components – Class 1 Components*” (3), Section III, Division 5, “*Rules for Construction of Nuclear Facility Components – High-Temperature Reactors* (4),” and Section II, “*Materials* (5).”

3.0 ASME CODE DATA REQUIREMENTS

Section III, Division 1, Subsection NB of the ASME Code (3) provides the material data for reactor components subjected to temperatures below 700°F and 800°F for ferritic and austenitic steels, respectively. Reactor vessel components and fuel clad/duct materials whose respective temperatures are below these limits do not require design/analysis for creep, creep-fatigue, or creep-rupture. Below these limiting temperatures of 700°F and 800°F for

ferritic and austenitic steels, material behavior is considered *time-independent*, such that prolonged exposure times at these temperatures will not appreciably degrade the material behavior.

For temperatures above 700°F and 800°F for ferritic and austenitic steels, respectively, material behavior is considered *time-dependent*, where degradation mechanisms such as creep and creep-rupture play an important role. The time-dependent material behavior implies that the exposure time at a given elevated temperature promotes creep and creep-rupture under load-controlled conditions and stress relaxation under deformation-controlled conditions. As such, the time-dependent material behavior is critical to the design life of a component. In general, the following set of test data categories as a function of temperature are required for an ASME Sec. III design:

Test Data Category	Function/Purpose
Physical	Common physical properties vs temperature.
Tensile	Yield, ultimate, %E, %RA and monotonic curves.
Creep	Determine strength reduction factors; Establish creep regimes and mechanisms; Support design for long-term service; creep rupture.
Stress Relaxation	Strain ratcheting; Supports material modeling.
Cyclic	Fatigue and creep-fatigue; Support design curves for ASME Code. Stress-strain curves; Determine non-isothermal response.
Fracture Toughness	Characterize embrittlement effects due to aging.
Multiaxial	Model component stress states. Verify von Mises assumption.

4.0 STRATEGIC ENDEAVORS

This section provides the necessary activities to support a long-term plan in the development and approval of an ASME Code Case for HT-9. In particular, the strategic plan will involve the following major topics:

1. HT-9 material database development and collection,
2. Developing necessary minimum mechanical properties for current ASTM A771 spec, and re-instatement of ASTM A831,
3. Obtain and maintain support and collaboration with commercial reactor module designer/vendor, (e.g., Gen4 Energy, Inc.)
4. Technical support and material supply chain from commercial material manufacturers, (e.g., Veridiam)
5. Maintain and enhance ASTM support for development of further product-form material specifications, and
6. Obtain expertise from national laboratory staff to assist with ASME Code Case development and material testing needs, (e.g., S. Sham, ORNL, K. Pasamehmetoglu, INL).
7. Continue ASME involvement with Sec. II and Sec. III
8. Development of a material research and development plan for HT-9
9. Develop a DOE/NE funding plan for material characterization

4.1 Precursors

Items #1 and #2 above are considered precursor activities that require completion before starting major activities.

4.1.1 HT-9 Material Database Collection

The Gap Analysis report developed by Brown et al. (1) identified available and sufficient, as well as unavailable or insufficient HT-9 material characterization data for mechanical, physical, and tensile properties. Nonetheless, collection of these data must be preserved in an electronic database environment, along with test records if available, akin to the Gen IV Materials Handbook database.

Each material property record should state the chemistry of testing specimen, test temperature or temperature range, material heat(s) tested, and reference test report. The archival database will eventually contain the necessary information for a formal ASME Code submittal.

4.1.2 Finalize ASTM A771 Update

A key activity during FY2012 was the re-instatement of ASTM Specification A771, “*Specification For Seamless Austenitic and Martensitic Stainless Steel Tubing for Liquid Metal-Cooled Reactor Core Components*” (6). ASTM subcommittee A01.10, *Stainless and Alloy Steel Tubular Products*, unanimous approved acceptance and adoption of the re-instated specification.

Another important feature of the currently re-instated ASTM A771 Specification is that the spec requires updating to current 2012 standards and must include minimum material mechanical properties at room temperature such as – Yield, Ultimate, % Elongation, % Reduction of Area, and Charpy V-Notch toughness or Plane Strain Fracture Toughness. In order to pursue updating of the ASTM specification for tubing, the material chemistry as currently stated for HT-9 will need to be maintained throughout all future product-forms. Material chemistry as currently specified for HT-9 is shown in Table 1 with the Unified Numbering System designation UNS S42100 or American Iron and Steel Institute designation, AISI 421.

Table 1 – ASTM A771 Alloy Composition Limits for Austenitic Stainless Steel Tubing

Grade UNS Designation	TP 316 S31600	... S38660	... S42100
Element	Weight, %		
Carbon	0.040–0.060	0.030–0.050	0.17–0.23
Manganese	1.00–2.00	1.65–2.35	0.40–0.70
Phosphorus, max	0.040	0.040	0.040
Sulfur, max	0.010	0.010	0.010
Silicon	0.50–0.75	0.50–1.00	0.20–0.30
Nickel	13.0–14.0	14.5–16.5	0.30–0.80
Chromium	17.0–18.0	12.5–14.5	11.0–12.5
Molybdenum	2.00–3.00	1.50–2.50	0.80–1.20
Titanium	...	0.10–0.40 ^A	...
Columbium, max	0.050	0.050	0.050 max
Tantalum, max	0.020	0.020	...
Tungsten	0.40–0.60
Nitrogen, max	0.010	0.005	...
Aluminum, max	0.050	0.050	0.050
Arsenic, max	0.030	0.030	...
Boron, max	0.0020	0.0020	...
Cobalt, max	0.050	0.050	...
Copper, max	0.04	0.04	...
Vanadium, max	0.05	0.05	0.25–0.35

^A Aim for 0.25.

Minimum mechanical properties may technically be justified from material certification data sheets (CERTS) obtained in accordance with ASTM A771. Additionally, various data has been collected in the AFCI Handbook (7), derived from several sources including

SANDVIK, FFTF and STIP-II irradiations and the NSMH. Also, Laboratory heats have been produced at LANL for research studies on carbide formation and phase transformation in HT-9 steel. Significant results obtained from laboratory heat testing could be used as a base for elevated engineering application purposes (35). Table 2 shows a compilation of mechanical properties.

Table 2 – Typical Mechanical Properties Derived from Several Sources

Source	Yield (MPa)	Ultimate (MPa)	%Elong.	%RA	Notes	Ref.
NSMH	620	745	10		Avg.	(2)
SANDVIK	490	700-850	---		Minimum	(7)
FFTF irradiations	620	730	7			(7)
STIP-II irradiations	633	885	11.7			(7)
CERTS-1	560	758	22.7	58.8	Tubing (A771)	(9)
CERTS-2	530	740	22	56	Hot-rolled plate (A831)	(10)

Thus, updating the current ASTM A771 specification for tubular products to include minimum required tensile mechanical properties should be a major focus in early FY2013. Additionally, ASTM withdrew specification ASTM A831, “*Standard Specification for Austenitic and Martensitic Stainless Steel Bars, Billets, and Forgings for Liquid Metal Cooled Reactor Core Components*,” in 2005. Along with updating ASTM A771, spec A831 for bars, billets and forgings should be pursued for re-instatement at the earliest convenience. The activity is necessary to initiate the process of ordering material from commercial material manufacturers and begin material characterization.

4.2 Commercial Reactor Designer Collaboration

Pre-conceptual studies of several advanced reactor systems have retained HT-9 steel as a material of choice for clad, components and piping. Commercial industry needs for advanced reactor systems, as well as DOE/NE support of relevant LMFR activities, represent a major driving force behind HT-9 development. Throughout FY2011 and FY2012, LANL engineering staff has been collaborating with Gen4 Energy, Inc. in design conceptualization of the LMFR reactor module, previously known as Hyperion, and currently termed the Gen4 Energy module.

Design decisions on use of HT-9 for reactor pressure vessel (RPV), coolant lines, valves, fittings and other components should be driven by the commercial reactor vendor/designer. Close collaboration with the commercial reactor designer is essential and recommended.

4.3 Commercial Material Manufacturer Collaboration

Material manufacturing support is a critical activity, that requires starting early in the process to allow for technical exchange in discussions for developing test samples for mechanical and physical properties characterization. One possible vendor is Veridiam, from San Diego, CA, which was previously known as Carpenter Special Products Corporation and owned by Carpenter Technology Corporation (CTC). The company maintains its focus on materials for the commercial nuclear industry, and originally supplied Westinghouse Hanford Corporation (WHC) with HT-9 tubing for FFTF applications. Manufacturers such as Veridiam could be interested in initiating production of HT-9 tubing if a supply of large amount of material is needed.

Additionally, it may become inconvenient to solely rely on a single material manufacturer to supply a future testing program. In certain instances, round-robin mechanical testing is conducted with material specimen supplied by different material manufacturers and testing performed by different vendors to achieve broad base and un-biased result. Therefore, it is recommended that several material manufacturers be requested to participate in a near-future program.

4.4 ASTM International Support

Although there is only one product-form specification (i.e., tubing) currently available for HT-9, there will likely be several other forms required for adoption as an ASME Code material, depending upon requirements developed by the commercial nuclear reactor owner/designer. That is, typical ASME Code product-forms for design of nuclear reactor pressure vessel and components, which require separate ASTM specifications are:

- Piping – Welded or seamless
- Tubing – Seamless
- Bar – Square or round
- Plate – Thickness variability
- Fittings – Valves, elbows, tees, etc.
- Forging – Pipe, fittings, etc.
- Casting
- Weld

Although a single chemistry will be utilized for all HT-9 material, separate ASTM specifications are required for each product-form. Because HT-9 product-forms are considered F/M stainless steels, the governing body within ASTM is Technical Committee A01, *Steel, Stainless Steel and Related Alloys*. This main committee oversees all other product-form specifications including inspection and test methods particular to material chemistry. As a minimum, the following subcommittees would need to be involved in the process of developing, accepting and approving further product-form specifications.

Table 3 – ASTM Subcommittees Responsible for Product-Form

S/C Identifier	S/C Name
<u>A01.06</u>	Steel Forgings and Billets
<u>A01.06.01</u>	Power Generation and Pressure Vessel Forgings
<u>A01.06.02</u>	General Industrial Forging
<u>A01.06.03</u>	Test Methods and Recommended Practices
<u>A01.10</u>	Stainless and Alloy Steel Tubular Products
<u>A01.11</u>	Steel Plates for Boilers and Pressure Vessels
<u>A01.13</u>	Mechanical and Chemical Testing and Processing Methods of Steel Products and Processes
<u>A01.15</u>	Bars
<u>A01.18</u>	Castings
<u>A01.22</u>	Steel Forgings and Wrought Fittings for Piping Applications and Bolting Materials for Piping and Special Purpose Applications
<u>A01.22.01</u>	Bolting
<u>A01.22.02</u>	Welding Fittings
<u>A01.22.03</u>	Forgings

Although ASTM specification A831, “*Standard Specification for Austenitic and Martensitic Stainless Steel Bars, Billets, and Forgings for Liquid Metal Cooled Reactor Core Components*,” for the HT-9 product-form described was withdrawn in 2005, limited effort may be employed to re-instate this specification, yet ASTM subcommittee A01.22 would need to approve/adopt A831 for re-instatement.

4.5 National Laboratories Support

National laboratories maintain the subject matter expertise on material science and engineering. LANL is leading the high dose irradiated materials testing (7). Within DOE Work Package, LANL is helping bridge the gap between the existing database for HT-9 steel and ASME requirements (1). Oak Ridge National Laboratory (ORNL) has played a key role in overseeing an ASME Code Case development for Alloy 617 to be used in Next Generation

Nuclear Plant (NGNP), Intermediate Heat Exchanger. Dr. Sam Sham, ORNL staff scientist has been involved with this project for over 10-years and maintains much of the corporate knowledge on the strategic activities required for a successful implementation. The services of knowledgeable individuals such as Dr. Sham should be encouraged and exploited to gain advantage in ensuring tactical and strategic success.

Furthermore, Dr. Sham and colleagues from Idaho National Laboratory (INL), as well as ASME Code consultants, published under DOE/NE support, the *Next Generation Nuclear Plant Intermediate Heat Exchanger Materials Research and Development Plan* (11), specifically for Alloy 617. This document provides not only a thorough descriptive listing of material properties and behavior characterization required for Alloy 617, but an estimated cost of specimen manufacture, testing and analysis for conducting such an endeavor. Additionally, the authors have collected available material behavior characterization data from known commercial heats of material from Huntington Alloys, Inc., ORNL, and GE-HTGR, in particular preserving all test records in the Gen IV Materials Handbook. Unfortunately, the case for HT-9 is not that clear-cut since much of the material behavior characterization data, as reported by Brown et al. (1), is not from commercial heats, or contains insufficient numbers of material heats for an ASME approved characterization of properties.

4.6 ASME Collaborations

Continuing ASME collaborations with Section II, *Materials*, and Section III, Division 1, *Nuclear Vessels* and Division 5, *High-Temperature Reactors*, is crucial to fostering the working relationship with Code bodies that ultimately will pass judgment on the material Code Case. Many of the Code authors are current staff members from national laboratories working on similar issues, such as ORNL and INL. Understanding rules, guidance and procedures for moving the Code Case forward through separate ASME committees and subcommittees is a time-consuming endeavor, but especially crucial from the standpoint of ensuring that all subcommittee member's comments and issues are addressed early and effectively. Because the ASME Code committees meet quarterly, if certain issues are not resolved in a timely fashion, these items may be deferred for 6-months or longer.

As stated in Section 4.5, the Alloy 617 ASME material Code Case has been in effect for over 10-years. The extended development time was partly caused by a number of issues including the significant elevated operation temperatures involved. A detailed assessment of HT-9 steel readiness status is needed, but the initial technical basis suggests that if DOE/NE support for HT-9 Code Case development continuation is provided a shorter length of time for approval and adoption of HT-9 could be envisioned.

4.7 Material Research and Development Plan

Section 4.5 described on-going ASME Code Case development for Alloy 617 (11) based upon a material research and development plan developed by INL. A similar research and development document specific to HT-9 should be completed, identifying the following initial needs based on the Gap Analysis by Brown et al. (1):

- Material properties
 - Relaxation strength
 - Isochronous stress-strain curves
 - Bi-axial stress rupture
 - Fatigue curve (S-N)
 - Cyclic stress-strain curve
 - Monotonic stress-strain curve
 - Young's modulus
 - Poisson's ratio
 - % Elongation
 - % Reduction of Area
- Physical properties
 - Density
 - Thermal conductivity
 - Thermal diffusivity
- Number of specimens
- Temperature range
- Coolant compatibility; corrosion/erosion effects

Results of the Gap Analysis showed potentially sufficient material characterization information for the following properties, although it was unclear how many commercial heats were available:

- Creep
- Creep fatigue
- Creep rupture
- Thermal expansion coefficient
- Yield strength
- Ultimate strength
- Fracture toughness, (K_{Ic})
- Impact toughness, (CVN)
- Low-cycles fatigue (S-N)
- Fatigue crack growth rate, (da/dN)

Results of our literature review also showed that HT-9 has been considerably tested with irradiation experiments performed in EBR-II and FFTF fast reactors, and irradiation test facilities such as HFIR. In general, property data has been gathered in the range of temperatures between 350-550°C, with few data points below 200°C and above 600°C, and a range of neutron doses that lies mainly between a couple of dpa and 150 dpa, with few data points above 200 dpa. Under the Advanced fuels research in the Fuel Cycle Research and Development program more data is being collected from irradiation temperatures up to 700°C and future irradiations are being performed to obtain irradiation data from doses >200 dpa.

Through these investigations a substantial irradiation effects database on mechanical properties has been developed (12),(13). It has been shown that HT-9 has excellent swelling resistance to doses above 200 dpa (14) and good creep properties.

Also, tensile and fracture properties for this steel have been investigated in detail for HT-9 steel to high dpa levels (15)(16)(17)(18) with recent tensile testing performed at 25 °C and at ~400°C on HT-9 after irradiation in FFTF to up to ~70 dpa at 373–433°C (19) that confirmed previous findings showing the significant effect irradiation temperature has on HT-9 hardening, in particular at low temperature below 400°C.

One of the main recommendations that come out from this literature search is the need to populate HT-9 mechanical properties database for fracture toughness and impact properties at low temperatures and high doses. More importantly, verification of pedigree relative to commercial heats will be necessary for all collected material characteristic and behavior data. As stated in Brown et al. (1), ASME Code Case material data for HT-9 requires 3-separate commercial heats of HT-9 material in the product-form and specification that will be utilized in-service. Although much of the open-source data collected to-date is beneficial in understanding HT-9 behavior, if the specimen were not derived from commercial heats, then it will not be approved by the ASME Code.

In the following topical areas, a brief summary is given on the temperature and irradiation dose ranges where data is available in the literature: tensile, impact, fracture toughness, and irradiation creep and swelling.

- **Tensile data:** There is a wealth of tensile property data gathered for HT-9 steel for temperatures in the range of 300°C to 550°C with doses up to 70 dpa. Tensile properties were determined for 12Cr-1MoVW after irradiation in the Experimental Breeder Reactor-II (EBR-II) at 390, 450, 500 and 550°C to ~13 dpa (20) and 23-25 dpa (21). Recently, tensile testing has been performed at 25 and at ~400°C on HT-9 after irradiation in FFTF to up to 70 dpa at 373–433°C (19). HT-9 tensile response to

irradiation was also reported after irradiations such as those performed in the High Flux Isotope Reactor (HFIR) at 90°C and 250°C to neutron doses of 1.5-2.5dpa (22) or the high flux reactor (HFR) in Petten (Netherlands) at temperatures in the range of 70°C to 370°C and damage dose levels up to 3 dpa (23) or more recently in spallation environments (19)(24).

- **Impact data:** Experimental data on DBTT shifts is found for HT-9 irradiations at temperatures above 200°C and dose values below 110 dpa. There is very few data at low temperature, with a few data points at 200°C for doses of a few dpa (25)(13). As mentioned above data at low irradiation temperatures is crucial, since low temperature embrittlement could result in steel failures during off-normal events. The reader is referred to (26) for a brief literature review of HT-9 impact properties under irradiation.
- **Fracture toughness:** HT-9 fracture toughness has been reported for irradiations performed at temperatures in the range 90-600°C. Irradiation doses have reached in some cases 180 dpa. HT-9 fracture toughness data has been reported (27), (28) after irradiation in the Experimental Breeder Reactor (EBR-II) to different doses in the range from 12 to 108 dpa, and different irradiation temperatures in the range from 290°C to 600°C and compared with HT-9 fracture toughness in the non-irradiated condition (29). Also, HT-9 compact tension specimens were irradiated in the Fast Flux Test Facility (FFTF) to about 180 dpa and irradiation temperatures in the range 405°C to 550°C (30). HT-9 fracture toughness has been also derived from irradiations in the High Flux Isotope Reactor (HFIR) at 90°C and 250°C to neutron doses of 1.5-2.5dpa (22). From the literature review, it is concluded that at temperatures typical of fast reactor operation, HT-9 fracture toughness could be adequate to exposures on the order of 100 dpa.
- **Irradiation creep and swelling:** There is abundant literature available on creep and swelling resistance of HT-9 with results derived from irradiations performed at temperatures in the range of 400-600°C, and maximum dose of 208 dpa (31), (14), (32). As said before, HT-9 shows very good properties of creep and swelling resistance. At the maximum swelling temperature of ~ 400–420°C, less than 2% swelling was observed for HT-9 irradiated to 200 dpa in the Fast Flux Test Facility (FFTF) (33). Concerning creep data, it must be noted that creep-rupture data is generally obtained for short times and small stresses. Extrapolation of rupture life from this data to the range of low stresses and long rupture times will be conservative; i.e. it will over estimate the actual value (33). Note that HT-9 thermal creep performance is being examined to qualify design models for thermal creep strain limits (34).

5.0 DOE/NE FUNDING PLAN

In order to continue the progress made in FY2012, LANL's proposed strategic activities specified in Section 4.1 through 4.7 require DOE funding support. Based on the necessary completion of precursor activities and development of an HT-9 specific database, as well as continuing interactions with ASME, ASTM and reactor designer (e.g., Gen4 Energy, Inc.), the following estimated funding levels are necessary through FY2013 and FY2014.

Item	Activity	FY2013 Man-Hours
1	HT-9 material database development (on-going activity through FY2014 and beyond)	520
2	Develop necessary minimum mechanical properties for A771 and A831	120
3	Obtain/maintain support and collaboration with commercial reactor module designer/vendor (e.g., Gen4 Energy)	240
4	Technical support and material supply chain from commercial material manufacturers (e.g., Veridiam)	240
5	Maintain and enhance ASTM support for development of further product-form material specifications	240
6	Obtain expertise/support from national laboratory staff to assist with ASME Code Case development and material testing needs	720
7	Continue ASME involvement with Sec. II and Sec. III	360
8	Develop material research and development plan for HT-9	800
SUB-TOTAL		3240

The above effort equates to ~2 FTE with a portion of the funding being distributed to potentially ORNL in assisting with Item #6 above.

The ASTM and ASME commitments are based on preparing technical material on HT-9, attending meetings and briefing respective committees on progress, providing technical presentations and developing technical specifications. ASME committee meets on a

quarterly basis (4 meetings per year) and ASTM meets twice yearly. Thus, the funding commitment is subdivided into labor and travel.

The level of funding estimated will ensure satisfactory progress is made in FY2013 in continuing the ASME Code Case development for HT-9, and to commence preparations for a materials research and development plan outlining required material testing.

6.0 CONCLUSIONS AND RECOMMENDATIONS

The strategy for developing an ASME Code Case for HT-9 described in this report suggest possible tasks and recommendations for a path forward:

- Collect HT-9 material data and preservation in the most updated electronic database environment, including test records if available.
- Finalize efforts for updating ASTM 771 and re-instatement of A831
- Further pursue close collaboration with the commercial reactor designer
- Engage in collaboration with several material manufacturers to commercially develop test samples for mechanical and physical properties characterization
- Involve ASTM subcommittees in the process of developing, accepting and approving further HT-9 product-form specifications
- Continue strategic activities in National Laboratories to develop HT-9 ASME Code Case
- Continue ASME collaborations and foster working relationships with Code bodies
- Pursue a materials research and development plan for HT-9 following the needs initially identified in the Gap Analysis (1).

ACKNOWLEDGEMENT

The authors are grateful to Dr. Stuart Maloy of Los Alamos National Laboratory and Dr. Neil R. Brown of Sandia National Laboratories for their assistance and expert advice on HT-9 material properties, testing and analysis. This work was accomplished for the US Department of Energy, Office of Nuclear Energy.

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